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Comparison of P-T Limit Curves of OPR-1000 and VVER-1000 RPV based on ASME Code Rule

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***Abstract*— The mechanical properties of reactor pressure vessel (RPV) continuously changes after the start of NPP operation mainly due to radiation irradiation. One of the prominent effect is irradiation embrittlement and as time goes by the material becomes more brittle. Because of this embrittlement, the operation of reactor pressure vessel should be carefully maneuvered to avoid brittle fracture. Hence the rate of heatup and cooldown and associated pressure control, which frequently repeated in plant operation, need to follow certain operational guide lines called P-T limit curve. Depending on the temperature of reactor vessel, the internal pressure shall be limited to lower than certain level. For this reason, pressure and temperature during heatup and cooldown processes should follow allowed range of pressure and temperature to prevent brittle fracture. Of all pressure boundary component of primary coolant system of NPP, the welded region of reactor pressure vessel degrade most due to the radiation embrittlement. The P-T limit curves for OPR-1000 and VVER-1000 reactor pressure vessel was investigated and compared. For OPR1000 family of reactors including APR1400 reactor, P-T limit curves were constructed following rules defined in KEPIC code (which is based on the ASME Code) for safe operation NPP, whereas VVER type reactors, it was constructed according to the rules of Russian code. This study compares the P-T limit curves of OPR-1000 and that of VVER 1000 reactor vessel according to Appendix G of the ASME Boiler and Pressure Vessel Code Section XI. Some of the significant differences are the minimum boltup temperature and lowest service temperature (LST) of OPR-1000 is 21.1°C and 77.8°C and those for VVER-1000 is 16.74°C and 30.6°C respectively. The lowest core critical temperature for OPR-1000 is 77.8°C and that for VVER-1000 is 63.8°C and corresponding maximum allowable pressure for OPR-1000 is 500 psi while that for VVER-1000 1200 psi.**

Keywords*— Radiation embrittlement, Pressure temperature limit curve, prevention of nuclear reactor fracture, Operational limit of pressurized water cooled raector, OPR-1000 reactor P-T limit curve; VVER-1000 ractor P-T limit curve

I. INTRODUCTION

At high neutron fluences, material damage on microstructural grain boundary and element transmutation occurs causing swelling and embrittlement. The radiation embrittlement of reactor pressure vessel of great concern in the operation of NPP. Therefore strict guide lines are given for the transient operation of reactor such as heat-up or cool-down [1]. It is usually considered that welded region is more susceptible to radiation embrittlement since the weld material contain more trace element than base metal. Some of the trace element cause adversely to the structural integrity of RPV [2][3][4].

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At low temperature reactor vessel material might exhibit brittle behavior, and in order to prevent such event happening, pressure should be lowered so that the postulated crack at the vessel wall is within the limit of stress intensity value at the temperature. On the other hand too low a pressure may initiate nucleate boiling in the core which is also limiting factor since it also causes inhibit proper heat transfer to coolant and may cause damage to the fuel. Another consideration is that the closure head stud bolts are tightly fastened during operation and the bolting region is in higher stress condition than other part of reactor pressure vessel. So stud and surrounding region is much higher stress state and that is susceptible to low temperature brittle fracture. These are some major considerations in setting up pressure-temperature limit curves [5].

During startup and shutdown as well as reactor trip, all components in the reactor coolant system (RCS) have to withstand all operating loads of temperature and pressure due to NPP operation mode changes. In order to protect reactor pressure vessel from brittle fracture, pressure - temperature (P –T) limit curves is constructed to be used as operational guideline of reactor operation and maintain NPP within safe state. During the construction of P-T limit curves for OPR-1000 and VVER-1000 reactor vessels, the beltline region which is most critical part of reactor vessel due to the susceptibility of brittle fracture. VVER-1000 reactor is manufactured by beltline weld for form RPV. The fabrication step of VVER-1000 reactor is shown in Figure 1. Although OPR-1000 does not have welded part within core region, it is conservatively assumed that the belt-line region of reactor is welded connection of lower head and cylindrical shell [6][7].

II. IMPORTANT PARAMETERS OF OPR-1000 AND VVER-1000 REACTOR VESSELS

A. RPV MATERIAL

OPR-1000 and VVER-1000 RPV are made from ferritic steels, but their composition, standard designation, and fabrication technique are differential. For example OPR-1000 specified American Society of Mechanical Engineers ASME SA508 for vessel, meanwhile material of VVER pressure vessel is 15Kh2NMFA [1].

Figure 1 shows the fabrication process of OPR-1000 reactor vessel and VVER-1000 reactor vessel where the difference of irradiation sensitive core region of the tow reactor is shown. In the figure, the beltline is the region of the reactor vessel that directly surrounds the effective height of

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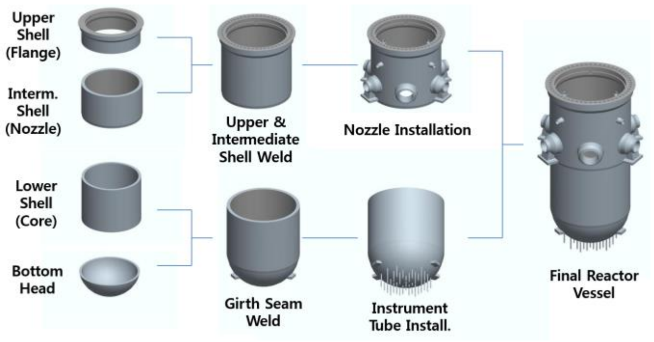
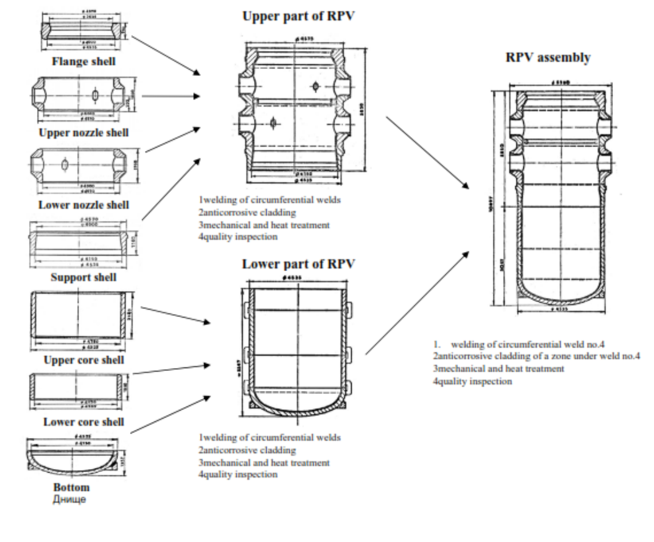
active core and corresponding to highest neutron fluence region [1].

Irradiation induced mechanical property changes of SA508 for OPR-1000 and 15Kh2NMFA in case of VVER-1000. The property change depends significantly upon the amount of residual elements present in the compositions of material such as copper, phosphorus, and nickel. Table 1 below shows typical value of chemical composition of SA-508 and 15Kh2NMFA. The impurity contents are slightly different from OPR-1000 to VVER-1000 RPV. It is note that the copper and phosphorus are considered in nil ductility temperature (NDT) evaluation and fracture toughness prediction in Russian code whereas US.NRC guide includes copper and nickel.



a) OPR1000 reactor vessel fabrication process and weld locations



b) VVER-1000 reactor vessel fabrication process and nozzle locations

Figure 1. Fabrication and assembly process of OPR1000 and VVER-1000 reactor vessel

Table 1 - Cu, Ni, and P constituents for OPR-1000 and VVER-1000 reactor vessels

Constituent	OPR reactor vessel base metal	OPR reactor vessel weld metal	VVER reactor vessel base metal	VVER reactor vessel weld metal
Cu	0.03	0.03	0.07	0.04
Ni	1	0.1	1.11	1.71
P	0.012	0.012	0.012	0.012

B. *NEUTRON ENERGY SPECTRUM AND NEUTRON FLUENCES*

The neutron exposure has to be determined from the surveillance specimens. The calculation is focused on the determination of the lead factor, which represents the ratio of exposure of the surveillance specimen to the highest anticipated exposure at the RPV wall. Fluence definition of VVER-1000 and OPR-1000 is different. For VVER's, neutrons with energies greater than 0.5 MeV are considered to form the fluence that is considered in vessel property change correlations. For PWR this energy boundary is taken to be 1 MeV. Table 2 shows design operating lifetime fluence for light water cooled reactors [8] [9].

TABLE 2. DESIGN OPERATING LIFETIME FLUENCE FOR LIGHT WATER COOLED REACTORS

Reactor type	Flux (n/m^2s) (E > 1 MeV)	Lifetime fluence (n/m^2) (E > 1 MeV)
VVER-440 core weld	1.2×10^{15}	1.1×10^{24}
VVER-440 maximum	1.5×10^{15}	1.6×10^{24}
VVER-1000	$3\text{-}4\times10^{14}$	3.7×10^{23}
PWR (OPR)	4×10^{14}	4×10^{23}
PWR (B&W)	1.2×10^{14}	1.2×10^{23}
BWR	4×10^{13}	4×10^{22}

C. *REGULATORY APPROACHES FOR ΔRT_{NDT}*

The national regulatory documents in connection with irradiation sensitivity of steels are defined in Regulatory Guide 1.99 Revision 2 in the USA, and OPR-1000 is regulated under similar rule that follows US regulations, and VVER-1000 reactor vessel is regulated under PNAE-G-7-002-86 in Russia. The main differences in regulatory approaches come from neutron spectrum considered and steel compositions. Russian code identifies copper and phosphorus whereas the US identifies copper and nickel as significant in analysis that will be basic to find chemistry factor for calculation of ΔRT_{NDT} [8][9].

Neutron fluence and chemical composition are the most relevant factors in radiation embrittlement. The general expression of ΔRT_{NDT} may be reduced to the following [8]:

$$\Delta RT_{NDT} = CF \times FF$$

Where CF is the chemical factor, and FF is the fluence factor (Φn) [8].

In the US, the fracture toughness curve is used to find equivalent temperature $T\text{-}\Delta RT_{NDT}$. ΔRT_{NDT} is the reference temperature for nil ductility transition and is defined in

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ASME code. The increase in ΔRT_{NDT} due to radiation exposure is evaluated according to the Code of Federal Regulations, Title 10, Part 50, Appendix G [8]. The predictive formula is defined in US Regulatory Guide 1.99 Revision 2 [8].

In VVER reactors, the neutron irradiation effects on mechanical properties are characterized by a shift of critical temperature of brittleness, T_k. The transition temperature, T_k in irradiated condition is determined directly from irradiated Charpy V-notch test results, using the regulatory definition of the transition temperature [9][10].

III. THE NORMAL OPERATION PRESSURE - TEMPERATURE LIMIT CURVE CONSTRUCTION PROCEDURE

For the safe normal operation of RPV, the operation should follow certain operation path call P-T curve. This curve is a result of consideration of material degradation due to radiation exposure of RPV beltline. The most provable failure mode fracture mode I, which is fast fracture failure due to tensile stress. This condition is expressed as follows [11][12]

$$\boldsymbol{K_{IC} \geq 2K_{IM} + K_{IT}} \quad (1)$$

Where $\boldsymbol{K_{IC}}$ = Reference critical stress intensity factor specified by Figure G-2210-1 [10]

K_{IM} = Stress intensity factor for membrane stress due to pressure, and is defined as $K_{IM} = \boldsymbol{\sigma}_m M_m$

M_m = Membrane correction factor defined in G-2214.1 to ASME Sec. XI [10]

$\boldsymbol{\sigma}_m = (Pr)/t$, membrane stress

P = Internal reactor vessel pressure, psia,

r = Inside reactor vessel radius, in.,

t = Reactor vessel wall thickness, in.

$\boldsymbol{K_{IT}}$ = Stress intensity factor for thermal stress defined in G-2214.3 to ASME Sec III [12].

The reference critical stress intensity factor is defined by Figure G-2210-1 of ASME Section III App. G and it is expressed in formula as below [11].

$$\boldsymbol{K_{IC} = 36.5 + 22.783e^{(0.036(T - RT_{NDT}))}} \ (\text{MPa}\sqrt{\text{m}}) \quad (2)$$

In equation (2), K_{IC} is a function of T of RT_{NDT}. The RT_{NDT} is determined by material chemical composition and accumulated radiation fluence. The right side of eq. (1) consists of the primary membrane stress and secondary membrane stress denoted by K_{IM} and K_{IT}. The $\boldsymbol{K_{IT}}$ is defined as follows

$$\boldsymbol{K_{IT}} = 0.579\times10^{-6}\times CR\times t^{2.5} \ (\text{MPa}\sqrt{\text{m}}) \quad (3\text{-a})$$

$$\boldsymbol{K_{IT}} = 0.458\times10^{-6}\times HU\times t^{2.5} \ (\text{MPa}\sqrt{\text{m}}) \quad (3\text{-b})$$

Where CR is cooldown rate in °C/ hr

HU is Heatup rate in °C / hr

$\boldsymbol{t}$ is the thickness of vessel wall in mm

The $\boldsymbol{K_{IT}}$ defined by eq. (3) corresponds to maximum temperature difference between vessel inner-surface to outer-surface throughout heatup or cooldown operation. Hence the intermediate value corresponding to temperatures along the operation shall be estimated proportionally. To do this transient conduction heat transfer calculation is need to find maximum through-wall temperature difference as well as the prorated value temperature difference along the temperature change. Hence $\boldsymbol{K_{IT}}$ is fully defined for input temperatures.

The K_{Im} is defined by $\sigma_m M_m$ where M_m is given in G-2214.1 to ASME Sec. III, appendix G as follows, for vessel thickness of 102 mm $\leq t \leq$ 305 mm [10]

- for an inside axial surface flaw: $M_m = 0.0293\sqrt{t}$
- for an outside axial surface flaw: $M_m = 0.0282\sqrt{t}$
- for an inside or an outside circumferential surface flaw: $M_m = 0.0140\sqrt{t}$

With this value, the remaining unknown is $\sigma_m = {}^{(Pr)}/_{t}$, since r is the ID of reactor vessel and t is the thickness of reactor vessel, the pressure is the variable need to be calculated.

IV. CALCULATION OF REFERENCE NIL-DUCTILITY TRANSITION TEMPERATURE RT_{NDT}

Since radiation degradation affects fracture toughness, it need to be properly accounted in the assessment of reference critical stress intensity factor. This critical stress intensity factor is determined from adjusted reference nil ductility transition temperature is defined in Regulatory Guide 1.99 (RG 1.99). The ART values of beltline material for weld metal and base metal at ¼ thickness and ¾ thickness are given by the following expression [11]:

$$\text{ART} = \text{Initial RT}_{\text{NDT}} + \Delta\text{RT}_{\text{NDT}} + \text{Margin} \quad (4)$$

Where

- Initial RT_{NDT} is the reference temperature for the unirradiated material.
- ΔRT_{NDT} is the measure of nil ductility transition temperature change during the reactor operation
- Margin is to compensate the uncertainties on both initial RT_{NDT} and $\Delta\text{RT}_{\text{NDT}}$.

A. The initial reference nil ductility transition temperature, Initial RT_{NDT}

The initial reference nil ductility transition temperature of APR1400 and OPR1000 reactor was determined by actual test of the material. VVER-1000 reactor's initial reference nil ductility transition temperature was assumed to be similar to that of OPR-1000 reactor vessel [8].

B. The shift of initial nil ductility transition temperature

ΔRT_{NDT} is the mean value of the adjustment in reference temperature caused by irradiation and is calculated as follows:

$$\Delta RT_{NDT} = (CF)f^{(0.28-0.10\text{lofg})} \quad (5)$$

The neutron fluence, f, at any depth in the vessel wall is calculated using methods that conform to the guide lines of RG 1.99 perform as follow [8]:

$$f = f_{surf}(\text{e}^{-0.24x}) \quad (6)$$

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In substituting the radiation fluence in eq. (6), the unit of 10^{19}n/cm^2 is used, and the fluence include those of fast neutron range (E>1 MeV). The calculated value of the neutron fluence at the inner wetted surface of the vessel at the location of the postulated defect is used in the calculation.

In eq. (5), CF stand for chemistry factor is defined RG 1.99. Base on the material used in reactor vessel beltline, the chemistry factor of copper (Cu) and nickel (Ni) is determined from table if RG 1.99. Since the material composition for reactor vessel and beltline weld are different, the CF for welds and vessel shell be determined separately.

C. Margin

According to RG 1.99, margin is quantity in temperature unit, *°C*, that is to be added to obtain conservative, upper-bound values of *ART* for the calculations required by Appendix G to 10 CFR Part 50 [8][9][13].

$$\text{Margin} = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2} \qquad (7)$$

Where

- σ_I is standard deviation for the initial RT_{NDT}.
- σ_Δ is the standard deviation of ΔRT_{NDT}.

V. CALCULATION OF THROUGH WALL TEMPERATURE GRADIENTS

P-T Limits curves for the core not critical and core critical operating condition apply for both the 1/4t and 3/4t locations. When combine pressure and thermal stress, it is usually necessary to evaluate stresses at the 1/4t location (inside surface flaw) and the 3/4t location (outside surface flaw) [12].

Analysis of heat up and cool-down rate from 10°C/hr to design limit of 100°F/hr intervals are performed by using the Finite Element Method (FEM) for thermal transient simulation, using ANSYS general purpose FEM analysis software. Locations at 1/4t and 3/4t are examined for the heat-up and cooldown transient. The temperature gradient at 1/4 and 3/4t wall depth for heatup and cooldown is obtained. The heat-up and cool-down rates vary from 10°C/hr to design limit of 100°F/hr. An example of thermal analysis result is shown in Figure 2 below [14].

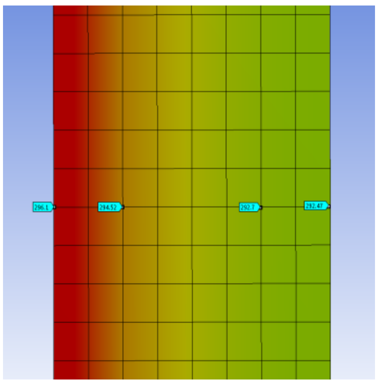

Figure 2. Output results for heatup rate at 37.8°C/hr

This calculation is necessary to accurately assess the K_{IT} in Eq. (3), where K_{IT} is maximum value. From the temperature transient, the maximum temperature difference between inner to outer surface corresponds to the Thermal Stress Intensity Factor given in Eq. (3). Hence at each temperature rise during heatup or temperature down during cooldown, the ratio from the maximum is calculated and corresponding value of Stress Intensity Factor is obtained.

VI. CALCULATION OF BELTLINE P-T LIMITS FOR HEATUP AND COOLDOWN OPERATION

For the beltline analysis, during normal operations, the following condition is maintained [12]:

$$K_{IC} = 2K_{Im} + K_{It} \qquad (8)$$

Where,

$$K_{Im} = M_m \times Pr_i/t \qquad (9)$$

$$K_{It} = \text{Max } K_{It} \times \Delta T / \text{Max } \Delta T \qquad (10)$$

$$K_{IC} = 36.5 + 22.783e^{[T-ART]} \qquad (11)$$

$$\text{Or } P = \frac{(K_{IC} - K_{IT})t}{2 * Mm * r} \qquad (12)$$

and

K_{Im}: Stress intensity factor for membrane stress due to pressure

K_{It}: Stress intensity factor for thermal stress

M_m: Membrane correction factor

P: internal reactor vessel pressure

r: inside reactor vessel radius

t: reactor vessel wall thickness

ΔT: Maximum steady-state temperature differential across the reactor vessel wall.

K_{IC}: Function of temperature, T, and adjusted reference temperature, ART, of the material at the crack tip (1/4t or 3/4t)

In eq. (8), the eq. (1) is modified to calculate the maximum allowable pressure for a given temperature. After substituting all variables, eq. (12) will give maximum allowable pressure. The overall procedure of obtaining P-T limit curve is shown in Figure 3. This calculation should be carried out for heatup and cooldown rate of 100°C, 80°C, 60°C, 40°C, 10°C, etc and hydrostatic test conditions.

VII. DETERMINATION OF OTHER TEMPERATURE LIMITS

A. Minimum Boltup Temperature of Reactor Vessel Closure Head

Base on ASME Code, Section III, Division I, Appendix G, Paragraph G-2222, the ASME Code requires that when the flange and adjacent shell region are stressed by the full bolt preload and by pressure shall not exceed 20% of the preoperational system hydrostatic test pressure [8][12].

This PT limit is a straight vertical line at the minimum boltup temperature ($T_{min\ boltup}$). The minimum metal temperature in the stressed region must be at least the initial RT_{NDT} plus any effects of irradiation.

$$T_{MIN\_BOLTUP} = ART + \text{Initial } RT_{NDT}$$

Since reactor flange where bolt connections is located is away from high radiation, the effect of irradiation embrittlement is negligible and therefore the T_{MIN_BOLTUP} is set to the same as Initial RT_{NDT}

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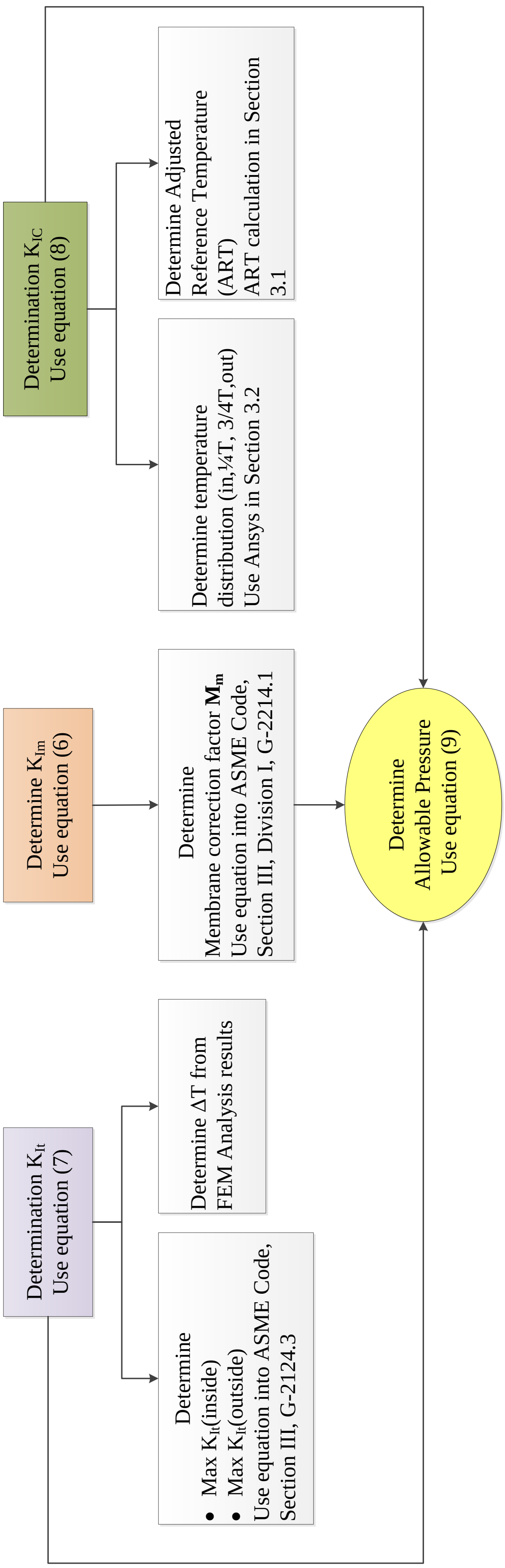

Figure 3. Process of allowable pressure determination

B. *Lowest Service Temperature (LST)*

The lowest service temperature (LST) is defined by paragraph NB-2332, Section III of the ASME Code, to be the minimum allowable temperature at pressures above 20% of the preoperational hydrostatic test pressure. This limit when enveloping all component, a more severe limit need to be applied and that is for clouse NB-2332, and is given by equation as follows [8]:

$$LST = RT_{NDT} + 37.8°C \text{ (or } 100°F)$$

C. *Minimum Pressure Below Lowest Service Temperature*

The minimum allowable pressure below the lowest service temperature is defined by NB-2332, Section III of the ASME Code, to be 20% of the pre-operation hydrostatic test [8].

$$P_{below\_LST} = 0.2P_{hydro\_test}$$

D. *Core Critical Curve*

The core critical curve is intended to provide additional margins of safety during core operation. The limitation is defined as 40°F above the minimum allowable temperature for heatup or cooldown and nor lower than the minimum temperature allowable for in-service hydrostatic test, in accordance with paragraph IV.A.3 in Appendix G to 10 CFR 50 [10].

E. *Hydrostatic Test Curves*

According to G-2400 in Appendix G to Section III of the ASME Code, Hydrostatic curve is to be performed by applying safety factor of 1.5 to the stress intensity factor for membrane stress due to pressure K_{Im} [12].

VIII. PRESSURE-TEMPERATURE LIMIT CURVE COMPARISON BETWEEN OPR-1000 AND VVER-1000 REACTOR VESSELS

A. *Pressure-Temperatue Limit Curves of OPR-1000 and VVER-1000 Reactor Vessels*

Construction of pressure-temperature limit curve at beltline of OPR-1000 and VVER-1000 RPV are performed for both heatup and cooldown condition for various temperature gradients. Combined with other temperature limits produces final pressure-temperature limits that should be followed during reactor operation. The results of pressure-temperature limit curve at beltline of OPR-1000 and VVER-1000 RPV are shown in Figure 4.

B. *Results and Evaluation*

- It showed quite a difference of shape and margin of VVER-1000 and OPR-1000 P-T limit curves. The main differences are come from differences of material, neutron flux, setpoint (hydrostatic test/normal operation pressure), thickness of reactor vessel wall. In order to eliminated computational error and construct accurate P-T limit curves, the time steps were set very small when doing the FEM analysis of temperature distribution through the RPV wall.

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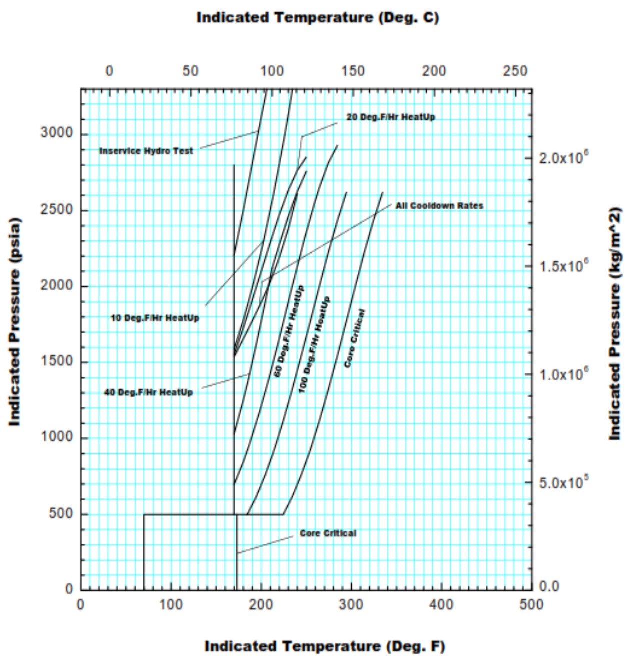
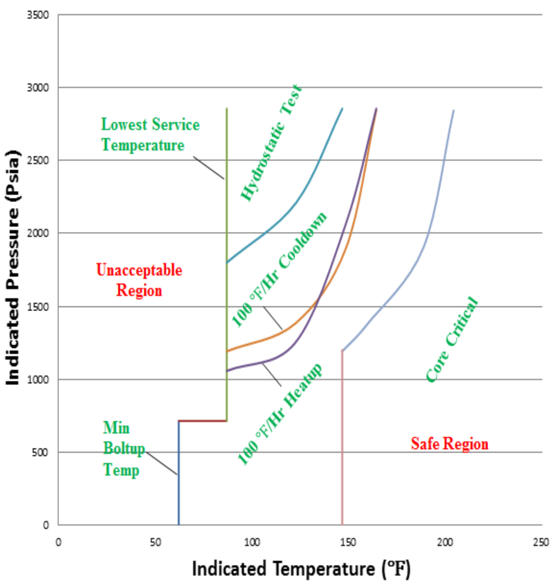


a) P-T Limits Curve of OPR-1000 RPV



b) P-T Limit Curve of VVER-1000 RPV

Figure 4. P-T limit curves of OPR-1000 and VVER-1000 RPV

- The difference of ART and Initial RT_{NDT} values of OPR-1000 and VVER-1000 are also causes difference for minimum boltup temperature and lowest service temperature (LST). Specific margin of minimum boltup temperature and lowest service temperature (LST) of VVER-1000 is 62.14°F and 87°F respectively whereas OPR-1000 is 70°F and 172°F.
- Hydrostatic pressure of VVER-1000 is less than OPR-1000 that leads difference of maximum pressure below lowest service temperature, specific value of VVER-1000 is 713.584 psia and OPR-1000 is 500 psia.
- Hydrostatic pressure also affect to hydrostatic test curve for both of VVER-1000 and OPR-1000 that have Safety factor of 1.5 is applied to stress intensity factor.
- The minimum boltup temperature is principally governed by Appendix G to Section III of the ASME code that requires the flange and adjacent shell region are stressed by the full bolt preload. Minimum boltup temperature of two RPV is quite similar, VVER-1000 is 62.14°F and OPR-1000 is 70°F.
- The Lowest service temperature is minimum allowable temperature at pressure above 20% of the preoperational hydrostatic test pressure. There are significant differences between LST of VVER-1000 (87°F) while LST of OPR-1000 (170°F).

One of the interesting point is that the lowest core critical temperature for VVER-1000 and OPR-1000 are 146.761°F and 172°F, but the maximum allowable pressure at that temperature is different where for VVER-1000 is 1200 psi and that for OPR-1000 is 500 psi.

IX.CONCLUSION

Pressure – Temperature (P-T) limit curve is generated to control the limit of permissible operating envelope for VVER 1000/320 RPV during reactor operation. P-T limit curves are also used by system designers in evaluating the consequences of transients which challenge reactor vessel integrity. Plant operating procedures guide operators to follow optimum heatup and cooldown curves that do not violate the allowable limits of P-T curve.

The P-T limit curves of the vessel of the belt-line region of VVER-1000 reactor were obtained for a 100°F/hr heatup and cooldown operation. The study was carried out by applying Appendix G of the ASME Boiler and Pressure Vessel Code, Section III, Appendix G to generate P-T limit curves of VVER 1000/320. To determine the adjusted reference temperature (ART) of the 15Kh2NMFA beltline material of VVER 1000/320, ΔRT_{NDT} was calculated based on Cu and Ni values and neutron fluence. For obtaining temperature through the thickness at inside surface, ¼t and 3/4t and outside surface of beltline region, FEM model of VVER 1000/320 was created and simulated by using transient thermal analysis module of ANSYS software 14.5.

Comparison of P-T limit curve for VVER-1000 and OPR-1000 was made. It showed that the minimum bolt-up temperature is quite similar but the minimum allowable LST showed significant difference, in which LST for VVER-1000 was 87°F while that for OPR-1000 was 170°F. The lowest core critical temperature also showed some difference where it for VVER-1000 is 146.70°F and it for OPR-1000 is 172°F, but the maximum allowable pressure at that temperature for VVER-1000 is 1200 psi while it for OPR-1000 is 500 psi.

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